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U. S. Nuclear Regulatory Commission
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South Texas Project
Units 1 & 2
Docket Nos. STN 50-498 & 50-499
Technical Specification Bases Change

In accordance with the South Texas Project Technical Specification Bases Control Program (Section 6.8.3.m) the attached pages are for your information and for updating the NRC copy of the Technical Specifications Bases. These changes are made to the following Technical Specification Bases pages:

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Note 1 Amendment 164/154 revise the Technical Specification Bases to establish a new programmatic requirement, largely performance-based framework for ensuring steam generator tube integrity.

Note 2 Amendments 161/151 revise the Technical Specification Bases to allow alternate acceptance criteria for Surveillance Requirement 4.7.7.e.3.

Note 3 Amendments 154/142 revise the Technical Specification Bases to remove information specific to Model E Steam Generators and references to Delta 94 Steam Generators.

Note 4 Amendments 152/140 revise the Technical Specification Bases to extend the interval between slave relay test in the engineered safety features actuation system instrumentation from 3 months to 18 months.

Note 5 Amendments 153/141 revise the Technical Specification Bases to eliminate shutdown actions associated with radiation monitoring instrumentation.

Note 6 Amendments 160/150 revise the Technical Specification Bases to relax restrictions on containment purge isolation valve operation.

Note 7 No amendment - Revised to add the function of P-4 interlock to close steam generator blowdown isolation valves upon a reactor trip when the source range is blocked.

Note 8 No Amendment - This change clarifies Table B.3.3.5-1 to state that electrical penetration space fans and battery room fans are included as required safety support systems for the remote shutdown system.

Note 9 Amendment 148/136 revised the Technical Specification Bases to allow automatic operation of the atmospheric steam relief valves during MODE 2 to maintain secondary side pressure at or below an indicated steam generator pressure of 1225 psig during startup and shutdown.

Note 10 Amendment 149/137 revise the Technical Specification Bases to delete the references to specific valves which must be secured to isolate potential uncontrolled dilution of boron in Mode 5 with loops unfilled and in Mode 6.

Note 11 Unit 2 Amendment 153 revises the Technical Specification bases for Unit 2 to alleviate the requirement to perform the quarterly operability surveillance on a Pressurizer Power Operated Relief Valve block valve that is being maintained closed.

Note 12 Amendments 156/144 revise the Technical Specification Bases to require that verification of Containment Spray System nozzle operability be performed only after maintenance that could result in nozzle obstruction.

Note 13 Amendments 146/134 - The Technical Specification bases is revised to better reflect the four train Auxiliary Feedwater System design at STP.

Note 14 No Amendment - This change includes an enhancement to Action Statement 3.8.1.1.d

Note 15 Amendment 163/152 revised the Technical Specification Bases to increase the allowed outage time for inoperable Remote Shutdown System components to a time that is more consistent with their safety significance.

Note 16 Administrative change to 3/4.8.1.1 to eliminate an incorrect reference to 4756 volts as a maximum operating voltage for 4000 volt motors.

If you have any questions on this matter, please contact Marilyn Kistler at (361) 972-8385 or me at (361) 972-7136.

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# INDEX

BASES

## 3/4.1 REACTIVITY CONTROL SYSTEMS

## BASES

### 3/4.1.1 BORATION CONTROL

#### 3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{avg}$. In MODES 1 and 2, the most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN OF 1.3%, $\Delta k/k$ is required to control the reactivity transient. The 1.3% $\Delta k/k$ SHUTDOWN MARGIN is the design basis minimum for the 14-foot fuel using silver-indium-cadmium and/or Hafnium control rods (Ref. FSAR Table 4.3-3). Accordingly, the SHUTDOWN MARGIN requirement for MODES 1 and 2 is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement of 1.3% $\Delta k/k$ is based on an uncontrolled RCS cooldown from a steamline break accident. The variable SHUTDOWN MARGIN requirement is based on the results of a boron dilution accident analysis, where the SHUTDOWN MARGIN is varied as a function of ARI N–1 Critical Boron Concentration, to guarantee a minimum of 15 minutes for operator action after a boron dilution alarm, prior to a loss of all SHUTDOWN MARGIN.

When SHUTDOWN MARGIN limits are not met the ACTION requires operators to initiate boration. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. The boron concentration source shall be greater than the required SHUTDOWN MARGIN boron concentration. Higher source boron concentration and higher flow rates will restore SHUTDOWN MARGIN quicker. The boration parameters of 30 gpm and 7000 ppm represent typical values when the borated water source is the boric acid tanks.

#### 3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

3/4.2.5 DNB PARAMETERS (continued)

instrument loops, the maximum indicated RCS average temperature should be equal to or less than 595 °F. The value for RCS flow rate is the average from a minimum of at least 2 flow transmitters per RCS loop using plant computer/QDPS points.

The value for thermal design RCS flow rate presented in Technical Specification 3.2.5 is an analytical limit. The minimum thermal design RCS flow rate is 392,000 gpm. To provide additional operating margin, a higher value for thermal design flow rate may be used if supported by cycle specific analysis. The minimum measured flow in the Core Operating Limits Report is the thermal design flow rate assumed for a particular cycle plus RCS flow measurement uncertainties. The RCS flow measurement uncertainty is 2.8% using the precision heat balance method or 2.1% using the elbow tap methods described in WCAP 15287, "RCS Flow Measurement for the South Texas Projects Using Elbow Tap Methodology", dated August, 1999. The elbow tap Dp measurement uncertainty presumes that elbow tap Dp measurements are obtained from either QDPS or the plant process computer. Based on instrument uncertainty assumptions, RCS flow measurements using either the precision heat balance or the elbow tap Dp measurement methods are to be performed at greater than or equal to 90% RTP at the beginning of a new fuel cycle.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

## 3/4.3 INSTRUMENTATION

## BASES

### 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," supplements to that report, WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and the South Texas Project probabilistic safety assessment (PSA). Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System instrumentation.

The 18-month slave relay test interval is based on information contained in WCAP-13878, Rev. 1, "Reliability Assessment of Potter & Brumfield MDR Series Relays." These assessments set conditions and provide guidance for maintaining the reliability necessary to continue 18-month testing.

ACTION 4 of Table 3.3-1 is modified to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SHUTDOWN MARGIN required by Technical Specifications. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes must also be evaluated to ensure they do not result in a loss of SHUTDOWN MARGIN. Control rod withdrawal is not allowed.

ACTION 5 of Table 3.3-1 for the Extended Range Neutron Flux Instrumentation is similar to ACTION 4 for the Source Range Instrumentation. The Action indicates that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SHUTDOWN MARGIN required by Technical Specifications. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient must also be evaluated to ensure they do not result in a loss of SHUTDOWN MARGIN. Control Rod withdrawal is not allowed.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

When control rods are at the top or above the active fuel region (≥ step 259), they are no longer capable of adding positive reactivity to the core, and as such, they are not capable of rod withdrawal as intended by MODE 5*. Therefore, ACTION 10 on Table 3.3-1 is not applicable in this region. This allows the Reactor Trip Breakers to be closed, without meeting the requirements of MODE 5*, while unlocking and stepping the control rods to a position no lower than 259. (CR 97-908-17)

Several ACTIONS in Tables 3.3-1 and 3.3-3 have been revised to change the allowed outage times and bypass test times in accordance with WCAP-10271 and WCAP-14333. Additionally, some ACTIONS have been divided such that only certain requirements apply depending on whether the Functional Units have been modified with installed bypass test capability.

Regardless of whether the Functional Units have installed bypass test capability, it should be noted that in certain situations, the ACTIONS permit continued operation (for limited periods of time) with less than the minimum number of channels specified in Tables 3.3-1 and 3.3-3. For example, Table 3.3-1 Functional Unit 11 (Pressurizer Pressure - High) requires a minimum of 3 channels operable. However, since continued operation with an inoperable channel is permitted beyond 72 hours, provided the inoperable channel is placed in trip, and since periodic surveillance testing of the other channels must continue to be performed, ACTION 6 permits a channel to be placed in bypass for up to 12 hours to permit testing. Thus, for a limited period of time (12 hours), 2 channels, or one less than the minimum, would be permitted to be inoperable.

Actuation relays consist of slave relays, including the relay contacts for actuating the ESF equipment. If a slave relay becomes inoperable for a particular component(s), then the associated component(s) LCO Required Action should be entered. If an entire train of slave relays for a functional unit becomes inoperable, then the Required Action for the functional unit actuation train should be entered. (CR 00-13604-7)

During a plant shutdown for refueling, the Normal Containment Purge System is in operation. The Supplementary Containment Purge System may be used during normal plant operation. Redundant Class 1E radiation monitors (i.e., the Reactor Containment Building [RCB] Purge Isolation) monitor the radiation in these purge lines. Upon either monitor sensing radiation above a preset limit, a signal is sent to the ESFAS logic trains, and the Containment ventilation isolation signal is actuated. In a LOCA, both Normal and Supplementary purge lines are isolated by a Safety Injection (SI) signal. Actuation of the purge isolation by these radiation monitors is not credited in the LOCA accident analyses, and is only a backup function for this event. The subject radiation monitors are credited for purge line isolation for a fuel handling accident.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

ACTION 18.a. applies when the actuation logic for RCB Purge Radioactivity – High is inoperable because it affects both channels. The required action is to maintain the isolation valves closed. Loss of power supply to the output ESF relays of either channel of these monitors will be considered inoperable actuation logic and the isolation valves will be maintained closed in accordance with proposed ACTION 18.a. This is because this failure mode will result in the inability of the other actuation signals to close the purge valves if the initial signal is reset.

In MODE 1, 2, 3, 4, or 5##, when one of the two required channels of RCB Purge Radioactivity – High is inoperable, ACTION 18.b.1 requires restoration within 30 days. The allowed outage time is a reasonable time for easily accessible non-risk-significant instrumentation. The required action is modified by a note that allows the supplementary purge valves to be opened in MODE 1-4 under administrative control during the 30-day allowed outage time to permit operation of the supplementary purge system for up to 2 hours at a time for the evolutions permitted by the Technical Specifications (containment pressure control, ALARA and respirable air quality needed for personnel entry into containment and for surveillance tests that required the valves to be open). The 2-hour allowance is adequate time for the routine pressure control purge operations during power operation. The note also allows the normal or supplementary purge supply and exhaust valves to be open up to 6 hours at a time in MODE 5## for required purge operations. The 6-hour duration is justified because the design basis event in this MODE would be expected to be a slower developing event and purge operations in support of refueling activities are typically much longer than those done at power. Opening the valves for purge operations is not permitted after the 30-day allowed outage time has expired.

In MODE 1 – 4, the safety analysis credits only the SI signal for actuation of CVI. As a backup, the operable radiation monitoring channel would still be available to actuate containment isolation. In MODE 5 and 6, there is no credible LOCA event and the design basis postulated event is a fuel handling accident in containment.

Administrative control during purge evolutions with an inoperable radiation monitoring channel would include the operator ability to manually initiate CVI from the control room handswitch and typically include an assessment of plant conditions for potential actuation precursors, monitoring containment radiation and limiting purge duration.

ACTION 18.b.2 applies in MODE 1, 2, 3, 4, and 5## when both channels of RCB Purge Radioactivity – High are inoperable. The action requires the purge isolation valves to be maintained closed and there is no provision for purge operation under administrative control.

ACTION 18.c. applies to the condition where one or both RCB Purge Radioactivity – High channels are inoperable in MODE 6 during movement of irradiated fuel or CORE ALTERATIONS. The ACTION directs the user to apply the requirements of TS 3/4.9.9 for an inoperable Containment Ventilation Isolation System during Refueling. With one inoperable channel of RCB Purge Radioactivity – High inoperable, the action includes a provision that allows purge operations for up to 6 hours at a time. The basis for the 6 hour duration of the purge is the same as described above for MODE 5##.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

ACTION 27 for an inoperable channel of control room ventilation requires the associated train of control room ventilation to be declared inoperable and the appropriate action take in accordance with Specification 3.7.7. Each control room ventilation system (train) is actuated by its own instrumentation channel. Consequently an inoperable channel of ventilation actuation instrumentation renders that system/train of ventilation inoperable and Specification 3.7.7 prescribes the appropriate action.

ACTION 28.a. provides 7 days to place the Control Room ventilation in the recirculation and make-up filtration mode of operation at 100% capacity (any two of the three trains of control room makeup and cleanup filtration meet the 100% capacity requirement) when one the two radioactivity-high actuation channels is inoperable. This time is acceptable because there is still an operable channel that will function to realign the control room envelope on a high radiation signal unless the failure mode is due to the output power supply. However, in that case, the operator can manually initiate the function. The 7 day allowed outage time is based on the low probability of a Design Basis Accident (DBA) occurring during this time period, and ability of the remaining train to provide the required capability.

ACTION 28.b. applies when both channels of control room ventilation radioactivity-high are inoperable and requires the ventilation system to be placed in recirculation and make-up filtration within 1 hour. The action includes an option to relax the time required to realign the Control Room Makeup and Cleanup Filtration System to the recirculation and makeup mode from 1 hour to 12 hours if the additional requirement to immediately suspend core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool is imposed. The option would apply in any of the four following scenarios: 1) plant operating and no fuel movement or crane operations over the spent fuel pool, 2) plant operating and fuel movement or crane operations being performed, 3) plant shutdown and fuel movement or crane operations being performed, and 4) plant shutdown and no fuel movement or crane operations being performed. (Although Scenario 4 would require no action other than logging the inoperability, it will preclude core alterations, fuel movement and crane movement with loads over the spent fuel pool). The additional restriction provides assurance that potential radiation releases from design basis accidents inside and outside containment have been considered for this configuration. The option permits fuel movement and crane operation with loads over the spent fuel pool if the Control Room makeup and cleanup filtration system is operating at 100% capacity (any two of the three 50% trains that comprise the system).

The option to allow a 12 hour action time is also consistent with TS 3.7.7 for Control Room HVAC, which allows all three trains of the HVAC to be inoperable for 12 hours.

ACTION 28.c. applies for MODE 1, 2, 3, & 4. It suspends core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool and requires the plant to be placed in a MODE where the Technical Specification does not apply.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

ACTION 28.d. applies in MODE 5 & 6 and requires the suspension of core alterations, movement of irradiated fuel assemblies and crane operations with loads over the spent fuel pool. This effectively precludes the design basis accidents that the control room radioactivity-high actuation system is designed to mitigate.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on $T_{avg}$ below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level and allows Safety Injection block so that components can be reset or tripped. Reactor tripped with the source range blocked provides a non-protective function that closes the Steam Generator Blowdown isolation valves and allows reopening the valves after the source range block is reset.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, reinstates steamline isolation on low compensated steamline pressure signals, and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, allows the manual block of steamline isolation on low compensated steamline pressure signals, and enables steam line isolation on high negative steam line pressure rate (when steamline pressure is manually blocked).

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 (NOT USED)

## 3/4.3.3.2 (Not Used)

## 3/4.3.3.3 (Not Used)

## 3/4.3.3.4 (Not Used)

## 3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

### BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG power operated relief valves (PORVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control at the remote shutdown panel, and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation Functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

### APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

## LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table B 3.3.5-1.

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (initial and long term),
- RCS pressure control,
- Decay heat removal via the AFW System and the SG safety valves or SG PORVs,
- RCS inventory control via charging flow, and
- Safety support systems for the above Functions, including service water, component cooling water, and onsite power, including the diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the Remote Shutdown System Function are OPERABLE. In some cases, Table B 3.3.5-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate instrumentation or control sources is OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

## APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

## ACTIONS

ACTION a. addresses the situation where one or more required Functions of the Remote Shutdown System is in a condition where one or more of its required channels are inoperable. This includes the control and transfer switches for any required Function. The Required Action is to restore the required Function to OPERABLE status within 30 days. The allowed outage time is based on operating experience and the low probability of an event that would require evacuation of the control room. If the Required Action and associated allowed outage time of ACTION a. is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed outage time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Action a. is modified by a note that states that separate condition entry is allowed for each function. The allowed outage time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

ACTION b. states that LCO 3.0.4 does not apply. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

SURVEILLANCE REQUIREMENTS

SR 4.3.3.5.1 requires performance of a CHANNEL CHECK once every 31 days to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels.

As specified in the Surveillance, a CHANNEL CHECK is only required for those channels, which are normally energized. The Frequency of 31 days is based upon operating experience, which demonstrates that channel failure is rare. The provision to limit the channel check in SR 4.3.3.5.1 to normally energized instrumentation is included because it is accepted in the Standard TS NUREG-1431. None of the STP remote shutdown instrumentation is normally de-energized. The provision is retained to provide flexibility for any future design change that would replace an instrument that is normally energized with one that is not.

SR 4.3.3.5.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the 18 month Frequency.

SR 4.3.3.5.3 requires a CHANNEL CALIBRATION, which is a complete check of the instrument loop and the sensor. The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

Table B 3.3.5-1

Remote Shutdown System Instrumentation and Controls

| FUNCTION | REQUIRED NUMBER OF CHANNELS |
|---|---|
| **1. Reactivity Control** | |
| a. Extended Range Startup Rate | 2 |
| b. Extended Range Neutron Flux Level | 2 |
| c. Reactor Trip Breaker Position Indication | 1 per trip breaker |
| **2. Reactor Coolant System (RCS) Pressure Control** | |
| a. RCS Wide Range/Extended Range Pressure | 2 |
| b. Pressurizer Power Operated Relief Valve (PORV) Control and Block Valve Control | 1[a] |
| **3. Decay Heat Removal via Steam Generators (SGs)** | |
| a. RCS Hot Leg Temperature (Wide Range) | 1 per loop in 3 loops[b] |
| b. RCS Cold Leg Temperature (Wide Range) | 1 per loop in 3 loops[b] |
| c. AFW Controls | 2[b] |
| d. AFW Storage Tank Level | 2 |
| e. Steam Line Pressure | 1 per steam line in 3 steam lines[b] |
| f. SG Level (Wide Range) | 1 per SG in 3 SGs[b] |
| g. AFW Flow | 1 per SG in 3 SGs[b] |
| h. Steam Generator Power Operated Relief Valves (SG PORVs) | 2[b] |
| **4. RCS Inventory Control** | |
| a. Pressurizer Level | 2 |
| b. Charging Pump Controls | 1 |
| c Letdown Isolation Valves | 1 |
| d. Reactor Head Vent Throttle Valve | 1 |
| e. Reactor Head Vent Isolation Valves | 1 pair |
| **5. Safety Support Systems** | |
| a. Boric Acid Transfer Pumps | 1 |
| b. Accumulator Discharge Isolation Valves and Power Lockouts | 3 |
| c. CCW Pumps and Heat Exchanger Outlet valves | 2 |
| d. ECW Pumps | 2 |
| e. EAB HVAC Fans, including Electrical Penetration Space Fans and Battery Room Fans | 2 |
| f. Reactor Containment Fan Coolers | 3 |

(a) Controls must be for PORV and block valves on same line.
(b) The instruments and control parameters must be in the same OPERABLE RCS loop/secondary loop.

## 3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The instrumentation listed in Table 3.3-10 corresponds to the Category 1 instrumentation for which selection, design, qualification and display criteria are described in Regulatory Guide 1.97, Revision 2.

Consistent with the requirements of NUREG-0737, an evaluation was made of the minimum number of valid core exit thermocouples necessary for measuring core cooling. The evaluation determined the complement of core exit thermocouples necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities, including incore effects of the radial decay power distributions, excore effects of condensate runback in the hot legs, and nonuniform inlet temperatures. Based on this evaluation, adequate core cooling is ensured with two valid core exit thermocouples channels per quadrant with two core exit thermocouples per required channel. The core exit thermocouple pair are oriented radially to permit evaluation of core radial decay power distribution. Core exit temperature is used to determine whether to terminate Safety Injection, if still in progress, or to reinitiate Safety Injection if it has been stopped. Core exit temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of core exit thermocouples are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of core exit thermocouples indicate the radial temperature gradient across their core

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

quadrant. The unit specific response to Item II.F.2 of NUREG-0737 further discusses the core exit thermocouples. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient. The subcooling margin monitor requirements are not affected by allowing 2 thermocouples/channel/quadrant as long as each channel has at least four operable thermocouples in any quadrant (e.g., A Train has four operable thermocouples in one of the quadrants, and C Train has four operable thermocouples in the same quadrant or any other quadrant.). This preserves the ability to withstand a single failure.

ACTION 39.a. allows 30 days for the restoration if one of the two channels of Containment Radiation-High is inoperable. If the instrument cannot be restored in the allowed 30 days, the required action is to submit a report to the NRC outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status. The 30 days and required action is acceptable based on operating experience, the low likelihood of an event requiring the function, the available redundant channel, and the pre-planned actions defined before loss of function.

ACTION 39.b. allows 7 days for restoration of at least one channel if both channels of Containment Radiation-High are inoperable. If a channel cannot be restored in the required time, the action is to submit a report as described above. The allowed outage time of 7 days is based on the relatively low probability of an event requiring instrument operation and the availability of alternate means to obtain the required information. Prompt restoration of at least one channel is expected because the alternate indications may not fully meet all performance qualification requirements applied to the instrumentation. Therefore, requiring restoration of one inoperable channel of the function limits the risk that the function will be in a degraded condition should an accident occur.

The pre-planned alternate monitoring capability is provided by a radiation monitor that will be temporarily placed outside the Reactor Containment Building. Implementation elements include the following:

- Procedure directing the temporary monitor to be installed within 7 days with one inoperable channel of Containment Radiation-High and within 72 hours with both channels of Containment Radiation-High inoperable.
- Requirements for the monitor and its installation on the platform.
- Procedure provisions for the alternate monitoring capability for core damage assessment, dose assessment, and accident classification.
- Calculational basis for relating the alternate monitor readings to accident conditions.

ACTION 40.a. requires restoration within 30 days if a channel of steam line radiation monitoring or steam generator blowdown line radiation monitoring is inoperable, provided there is functional diverse channel. If the channel cannot be restored in the 30 days, a report must be submitted to the NRC outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel to OPERABLE status. The steam line radiation monitor and the steam generator blowdown radiation monitor are considered to be functionally redundant to one another. The allowed outage time and required

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

action are acceptable based on operating experience, the low likelihood of an event requiring the function, the available functionally redundant channel, and the pre-planned actions defined before loss of function.

ACTION 40.b. requires restoration within 7 days if a channel of steam line radiation monitoring or steam generator blowdown line radiation monitoring is inoperable, and there is no functional diverse channel. If the channel cannot be restored in the 7 days, a report must be submitted to the NRC. The allowed outage time of 7 days is based on the relatively low probability of an event requiring instrument operation and the availability of alternate means to obtain the required information. Prompt restoration of the channel is expected because the alternate indications may not fully meet all performance qualification requirements applied to the instrumentation. Therefore, requiring restoration of one inoperable channel of the function limits the risk that the function will be in a degraded condition should an accident occur.

STP s procedure for monitoring primary to secondary leakage is the pre-planned alternate method that will be implemented for this ACTION.

3/4.3.3.7 (Not Used)

3/4.3.3.8 (Not Used)

3/4.3.3.9 (Not Used)

3/4.3.3.10 (Not Used)

3/4.3.3.11 (Not Used)

3/4.3.4 (Not Used)

3/4.3.5 ATMOSPHERIC STEAM RELIEF VALVE INSTRUMENTATION

The atmospheric steam relief valve manual controls must be OPERABLE in Modes 1, 2, 3, and 4 (Mode 4 when steam generators are being used for decay heat removal) to allow operator action needed for decay heat removal and safe cooldown in accordance with Branch Technical Position RSB 5-1.

The atmospheric steam relief valve automatic controls must be OPERABLE with a nominal setpoint of 1225 psig in Modes 1 and 2 because the safety analysis assumes automatic operation of the atmospheric steam relief valves with a nominal setpoint of 1225 psig with uncertainties for mitigation of the small break LOCA. In order to support startup and shutdown activities (including post-refueling low power physics testing), the atmospheric steam relief valves may be operated in manual and open, or in automatic operation, in Mode 2 to maintain the secondary side pressure at or below an indicated steam generator pressure of 1225 psig.

The uncertainties in the safety analysis assume a channel calibration on each atmospheric steam relief valve automatic actuation channel, including verification of automatic actuation at the nominal 1225 psig setpoint, every 18 months.

REACTOR COOLANT LOOPS and COOLANT CIRCULATION ( continued )

ACTIONs are provided with a similar requirement that, with no reactor coolant loop in operation, operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the required SHUTDOWN MARGIN are prohibited. Suspending the introduction into the RCS of coolant with boron concentration less than that required to meet the SHUTDOWN MARGIN limit is necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in reducing core reactivity below the required SHUTDOWN MARGIN limit.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

LCO 3.4.1.4.2.b

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 5 with the loops not filled and thus avoid a reduction in SHUTDOWN MARGIN.

BACKGROUND

During MODE 5 operations with the loops not filled, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position. The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 5 with the loops not filled, isolation of all unborated water sources prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 5 with the loops not filled is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves or mechanical joints during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves and mechanical joints are used to isolate unborated water sources. These devices have the potential to indirectly allow dilution of the RCS boron concentration in MODE 5. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 5 with the loops not filled.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

REACTOR COOLANT LOOPS and COOLANT CIRCULATION ( continued )

APPLICABILITY

In MODE 5 with the loops not filled, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

ACTIONS

The ACTIONS section allows separate ACTION entry for each unsecured unborated water source isolation valve or mechanical joint used for isolation.

Continuation of reactivity control activities is contingent upon maintaining the unit in compliance with this LCO. With any valve or mechanical joint used to isolate unborated water sources not secured in the closed position, all operations involving that could reduce the boron concentration of the RCS below the SHUTDOWN MARGIN must be suspended immediately. The Completion Time of "immediately" for performance of the required action shall not preclude completion of movement of a component to a safe position.

The required action to confirm the boron concentration is within limit is required to be completed whenever ACTION c. is entered.

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation devices secured closed. Securing the valves or mechanical joints in the closed position ensures that the devices cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve or mechanical joint and secure the isolation device in the closed position immediately. Once actions are initiated, they must be continued until the devices are secured in the closed position.

Due to the potential of having diluted the boron concentration of the reactor coolant, verification of boron concentration must be performed whenever ACTION c is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 4.4.1.4.2.2 These valves or mechanical joints are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 5 with the loops not filled is remote due to the fact that all unborated water sources are isolated, precluding a dilution. This Surveillance demonstrates that the devices are closed through a system walkdown.
The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the device opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.6
2. NUREG-0800, Section 15.4.6

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 504,950 lbs. per hour of saturated steam at the valve setpoint of 2500 psia.

RELIEF VALVES (Continued)

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Manual control allows a block valve to isolate a stuck-open PORV.

Unit 2 Only: An exception to Surveillance Requirement 4.4.4.2 is provided if the block valve is closed in accordance with the ACTIONS of TS 3.4.4. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV has already been declared inoperable.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.3.o, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.3.o, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.3.o. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY (Continued)

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT 1-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity. Refer to Action a. below.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.3.o and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY (Continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total from all SGs. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gpd. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Applicability

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the required ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the required

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY (Continued)

ACTIONS may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated required ACTIONS.

a. The condition applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, the plant must be shut down in accordance with the ACTION.

Seven days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, the ACTION statement allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This is acceptable since operation until the next inspection is supported by the operational assessment.

a. and b. Six hours to reach HOT STANDBY and an additional 30 hours to reach COLD SHUTDOWN are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

A condition monitoring assessment of the SG tubes is performed during SG inspections. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY (Continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.3.o contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

4.4.5.2 During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service (by plugging). The tube repair criteria delineated in Specification 6.8.3.o are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 and Reference 6 provide guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering MODE 4 following a SG inspection" ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. 10 CFR 50 Appendix A, GDC 19
3. 10 CFR 100
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976
6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines"

## 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

RCS Leakage Detection instrumentation consists of one Containment Atmosphere Radioactivity Monitor (gaseous or particulate), and the Containment Normal Sump Level and Flow Monitoring System.

The Containment Normal Sump Level and Flow Monitoring System leakage detection method is accomplished by monitoring the containment sump using two independent methods. One method is the Flow Monitoring System. This method measures the volume of water pumped out of the sump over a period of time and calculates an average leak rate. The other volumetric method involves measuring a change in the Containment Normal Sump Level over time, which also provides a means of manually or automatically calculating an average leak rate. Since both of these methods provide a means to detect average leak rate, they are redundant. OPERABILITY of the Containment Normal Sump Level and Flow Monitoring System is dependent on the operability of LI-7812 "Containment Normal Sump Level" **or** FQI-7823 "Containment Normal Sump Discharge."

### 3/4.4.6.2 OPERATIONAL LEAKAGE

#### Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the main steam safety valves and the majority is steamed to the condenser. The 1 gpm primary-to-secondary leakage safety analysis assumption is relatively inconsequential.

The SLB is more limiting for primary-to-secondary leakage. The safety analysis for the SLB assumes 500 gpd and 936 gpd primary-to-secondary leakage in the faulted and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was 1.0 μCi/gm DOSE EQUIVALENT 1-131 at a conservatively high letdown flow of 250 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFIED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpd per each steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 1). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

e. Reactor Coolant System Pressure Isolation Valve Leakage

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the Reactor Coolant Pressure Boundary are much lower, and further deterioration is much less likely.

b. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.

Surveillance Requirements

4.4.6.2.1 Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 states that this Surveillance Requirement is not required to be performed in until 12 hours after establishment of steady state operation.

Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity, containment normal sump inventory and discharge, and reactor head flange leakoff. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems."

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The Note in 4.4.6.2.1 states that this Surveillance Requirement is not applicable to primary-to-secondary leakage. This is because leakage of 150 gpd cannot be measured accurately by a RCS water inventory balance.

The 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

4.4.6.2.2 The Surveillance Requirements for Reactor Coolant System Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 1. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a Note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling. In MODES 3 and 4, the primary system radioactivity level may be very low, making it difficult to measure primary-to-secondary leakage. Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary-to-secondary leakage is less than or equal to 150 gpd through any one SG (Ref. 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 (Not Used)

CONTAINMENT VENTILATION SYSTEM (Continued)

fore, the SITE BOUNDARY dose guidelines of 10 CFR 100 would not be exceeded in the event of an accident during containment PURGING operation.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. Allowed leakage rates will be governed by the Containment Leakage Rate Program.

## 3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS – BASES

### 3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System both provide post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

Operability of the Containment Spray System is confirmed following maintenance activities that can result in obstruction of spray nozzle flow. Confirmation that the spray nozzles are unobstructed may be obtained by a visual inspection, or by an air or smoke flow test.

### 3/4.6.2.2 RECIRCULATION FLUID PH CONTROL SYSTEM

The operability of the recirculation fluid pH control system ensures that there is sufficient trisodium phosphate available in containment to guarantee a sump pH of ≥7.0 during the recirculation phase of a postulated LOCA. This pH level is required to reduce the potential for chloride induced stress corrosion of austenitic stainless steel and assure the retention of iodine in the recirculating fluid. The specified amount of TSP will result in a recirculation fluid pH between 7.0 and 9.5.

### 3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post LOCA conditions.

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1413.5 psig) of its design pressure of 1285 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

Five MSSVS, each with an orifice size of 16 $in^2$, are located on each main steam header, outside containment, upstream of the main steam isolation valves. The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 20.65 x $10^6$ lb/hr which is 120% of the total secondary steam flow of 17.20 X $10^6$ lb/hr for the steam generators at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$\mathrm{Hi}\,\Phi = \left(\frac{100}{Q}\right)\left(\frac{w_s h_{fg} N}{K}\right)$$

Where:

Hi Φ = Safety analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

K = Conversion Factor, 947.82 (BTU/sec)/MWt

$w_s$ = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lbm/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then $w_s$ should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then $w_s$ should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.

$h_{fg}$ = Heat of vaporization for steam at the highest MSSV operating pressure including allowances for tolerance, drift, and accumulation, as appropriate, Btu/lbm

N = Number of loops in the plant.

The calculated values are lowered an additional 9% full power to account for instrument and channel uncertainties.

## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater (AFW) System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

For planned motor-driven AFW pump (MDAFWP) out-of-service time exceeding 14 days, station procedures and application of the CRMP require that compensatory measures be implemented in accordance with the Probabilistic Safety Assessment modeling assumptions. Compensatory measures are to be implemented in accordance with the CRMP and plant procedures. These measures normally include the following:

- The work schedule contains no planned maintenance on required systems, subsystems, trains, components, and devices that depend on or that affect the remaining MDAFWP trains.
- The work schedule contains no planned maintenance activities in the switchyard that could directly cause a Loss of Offsite Power event. Maintenance activities identified after the Extended Allowed Outage Time (EAOT) begins that are required to ensure the continued reliability and availability of the offsite power sources, are permitted.
- If in Mode 1, 2, or 3, then verify the work schedule contains no planned maintenance on the turbine-driven AFW pump.
- The work schedule contains no planned maintenance that would result in the EW and the systems it supports being declared non-functional.
- The work schedule contains no planned maintenance that would result in an inoperable open containment penetration.
- The work schedule contains no planned maintenance on SWGR 1L(2L) or 1K(2K).
- The work schedule contains no planned maintenance on the 138 kV emergency transformer.
- The work schedule contains no planned EAOT for the SDG, EW, or EChW during the MDAFWP out-of-service period.

Should one or more of these compensatory measures not be met during the MDAFWP out-of-service period, action will be taken in accordance with the CRMP to restore the function. If indicated by the risk assessment conducted in accordance with the program, other actions may be taken by station management to reduce risk by restoration of other components, rescheduling work that might increase the risk, or placing the unit in a more appropriate configuration.

If entry into the Action is unplanned (i.e., a failure of a MDAFWP), station procedures require the implementation of the CRMP when it is determined that the out-of-service time will exceed 14 days. If one or more of the compensatory measures is not functional at the end of the 14 days, action will be taken in accordance with the CRMP to restore the function and to manage the risk.

AUXILIARY FEEDWATER SYSTEM (Continued)

If two MDAFWPs are inoperable (Action b), it is not necessary to restore both pumps to OPERABLE status within 72 hours. If one pump is restored to OPERABLE status, the plant is then in Action statement (a) and a different AOT applies (see CR 01-2103-14 for further details).

Each auxiliary feedwater pump is capable of delivering feedwater to the entrance of the steam generators with sufficient capacity to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation. Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle (Ref.: Calculations MC-5861 and ZC-7019). Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code. The AFW pumps are tested using the test line back to the AFST and the AFW isolation valves closed to prevent injection of cold water into the steam generators. This testing methodology confirms one point on the curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME Code, Section XI, satisfies this requirement. The STPEGS isolation valves are active valves required to open on an AFW actuation signal. Specification 4.7.1.2.1 requires these valves to be verified in the correct position.

3/4.7.1.3 AUXILIARY FEEDWATER STORAGE TANK (AFST)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with a MFWLB and failure of the AFW flow controller followed by a cooldown to 350°F at 25°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 (Not Used)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

The Control Room Makeup and Filtration System is comprised of three 50-percent redundant systems (trains) that share a common intake plenum and exhaust plenum. Each system/train is comprised of a makeup fan, a makeup filtration unit, a cleanup filtration unit, a cleanup fan, a control room air handling unit, a supply fan, a return fan, and associated ductwork and dampers. Two of the three 50% design capacity trains are required to be operable during the following modes of operation: shutdown, hot standby, normal operation, postulated accident condition, and loss of offsite power. The toilet/kitchen exhaust, heating, and computer room HVAC Subsystem associated with the Control Room Makeup and Filtration System are nonsafety-related and not required for operability.

The OPERABILITY of the Control Room Makeup and Cleanup Filtration System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 92-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

The ACTIONS specified during modes 5 and 6 with less than the minimum required Control Room Makeup and Cleanup Filtration Systems, or associated power systems, include suspending operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or refueling boron concentration necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SHUTDOWN MARGIN or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required SHUTDOWN MARGIN or refueling boron concentration limits. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

The time limits associated with the ACTIONS to restore an inoperable train to OPERABLE status are consistent with the redundancy and capability of the system and the low probability of a design basis accident while the affected train(s) is out of service. A limited allowed outage time of 12 hours is allowed for all three trains to be out of service simultaneously in recognition of the fact that there are common plenums and some maintenance or testing activities required opening or entry into these common plenums. This time is reasonable to diagnose, plan, and possibly repair problems with the boundary or the ventilation system. This is acceptable based on the low probability of a design basis event in that brief allowed outage time and because administrative controls impose compensatory actions that reduce the already small risk associated with being in the ACTION. The compensatory actions are consistent with the intent of GDC 19 to protect plant personnel from potential hazards such as radioactive contamination, smoke, and temperature, etc. Pre-planned measures should be available to address these concerns for intentional and unintentional entry into the condition. The compensatory actions include:

- Procedures will preclude intentionally removing multiple trains of Control Room Envelope HVAC from service if Containment Spray is not functional or intentionally making a train of Containment Spray unavailable when multiple trains of Control Room Envelope HVAC are out of service. For purposes of this compensatory action, Containment Spray is considered functional if at least one train can be manually or automatically initiated.

- The plant will not make planned simultaneous entries into TS 3.7.7 ACTION c. for MODES 1, 2, 3 and 4 and TS 3.7.8 ACTION b or d.

The compensatory action may include placing fans in pull-to-lock as necessary to preclude there being a motive force to transport contaminated air to a clean environment in the event of an accident. These compensatory actions also include administrative controls on opening plenums or other openings such that appropriate communication is established with the control room to assure timely closing of the system if necessary. Since the Control Room Envelope boundary integrity also affects operability of the overall system, entry and exit is administratively controlled. Administrative control of entry and exit through doors is performed by the person(s) entering or exiting the area. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.

Surveillance Requirement 4.7.7.e.3 verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the Control Room HVAC. During the emergency mode of operation, the Control Room HVAC is designed to pressurize the control room to at least 1/8 inch water gauge (in-wg) positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The Control Room HVAC is designed to maintain this positive pressure with two trains at a makeup flow rate of 2000 cfm. The frequency of 18 months is consistent with the guidance provided in NUREG-0800. If the surveillance results are less than 1/8 in-wg and the pressure differential is not positive, the surveillance requirement is considered not met and the appropriate action of TS 3.7.7 must be applied.

The surveillance includes a footnote allowing an evaluation of conditions where the differential pressure is positive but less than 1/8 in-wg. The measured positive relative pressure condition still assures that any leakage across this boundary location would be outleakage. Therefore, the functionality of the control room HVAC system is maintained with the degraded pressure condition within the envelope. The use of the footnote for a condition where the points are less than 1/8 in-wg is intended to be a temporary application until the points are restored to the design 1/8 in-wg in accordance with the corrective action program.

Compensatory actions may be applied based on the results of the evaluation provision of SR 4.7.7.e.3. The evaluation, including appropriate compensatory actions, must demonstrate that the dose limits of GDC 19 of Appendix A of 10CFR50 are met, including a 30 rem limit to the thyroid. If compensatory measures include self-contained breathing apparatus (SCBA) and potassium iodide (KI) tablets, then the requirements of Regulatory Position 2.7.3 of NRC Regulatory Guide, 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors" must be met.

The results of the evaluation for areas with differential pressure that is positive but less than 1/8 in-wg and the appropriate compensatory action are as described below.

STPNOC performs increased testing as a compensatory action to provide assurance against further degradation of the boundary and variances in pressure conditions external to the control room boundary.

1. Within 60 days of approval of the proposed Technical Specification change that provided for compensatory action in SR 4.7.7.e.3, and on a quarterly frequency thereafter, each train combination (e.g., A-B, B-C, or A-C) will be tested on a staggered test basis in the pressurization and recirculation cleanup mode of operation (i.e., the emergency mode).

2. If all test points for a train combination are greater than or equal to 1/8 in-wg positive relative to adjacent areas, that train combination will be removed from the increased testing. However, retest of the removed combination will be required if rebalancing is performed on one of the other train combinations.

To account for instrument uncertainties, a measurement is considered to be positive only if it is greater than the accuracy of the measurement process. The "nominal" 1/8 in-wg is acceptable for the Technical Specification surveillance.

During the period where increased testing is required because some test points are positive but are not greater than or equal to 1/8 in-wg positive pressure relative to adjacent areas, the following personnel protective compensatory measures will be in place.

1. For degraded conditions across walls with sealed penetrations where the likelihood of any inleakage condition resulting from changing conditions between testing would be minimal (i.e., a few cfm), potassium iodide (KI) tablets will be credited as the compensatory measure.

2. For degraded conditions across doors where seals have the potential for degradation and the inleakage condition resulting from changing conditions between testing would likely be more than minimal, self-contained breathing apparatus (SCBA) will be credited as the compensatory measure.

In both cases, crediting the compensatory action will only be required until the surveillance demonstrates that the differential pressure is greater than or equal to 1/8 in-wg. Conditions with positive differential pressure, but less than 1/8 in-wg will be addressed in the Corrective Action Program.

The procedural infrastructure to apply the compensatory actions is in place. KI is available to the Control Room crews and SCBA units are staged and ready for use by Control Room personnel. STP's emergency plan implementing procedures require that personnel radiation exposure in the control room be monitored so that appropriate personnel protective measures will be taken by the operators during accident conditions.

Reference: CREE 04-3148-34

## 3/4.7.8 FUEL HANDLING BUILDING EXHAUST AIR SYSTEM

The FHB exhaust air system is comprised of two independent exhaust air filter trains and three exhaust ventilation trains. Each of the three exhaust ventilation trains has a main exhaust fan, an exhaust booster fan, and associated dampers. The main exhaust fans share a common plenum and the exhaust booster fans share a common plenum. An OPERABLE ventilation exhaust train consists of any OPERABLE main exhaust fan, any OPERABLE exhaust booster fan, and appropriate dampers.

The OPERABILITY of the Fuel Handling Building Exhaust Air System ensures that radioactive materials leaking from the ECCS equipment within the FHB following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for the least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

The time limits associated with the ACTIONS to restore an inoperable train to OPERABLE status are consistent with the redundancy and capability of the system and the low probability of a design basis accident while the affected train(s) is out of service. The allowed outage time for one train of FHB exhaust ventilation or one exhaust filtration train being inoperable, or a combination of an inoperable exhaust ventilation train and an inoperable exhaust filtration train is 7 days. With more than one inoperable train of either FHB exhaust filtration or FHB exhaust ventilation, or with combinations involving more than one inoperable train of either the exhaust ventilation or the exhaust filtration, the allowed outage time is 12 hours. A limited allowed outage time of 12 hours is allowed for multiple trains to be out of service simultaneously in recognition of the fact that there are common plenums and some maintenance or testing activities required opening or entry into these common plenums. This time is reasonable to diagnose, plan, and possibly repair problems with the boundary or the ventilation system. This is acceptable based on the low probability of a design basis event in that brief allowed outage time and because administrative controls impose compensatory actions that reduce the already small risk associated with being in the ACTION.

The compensatory actions are consistent with the intent of GDC 19, GDC 60 and Part 100 to protect plant personnel and the public from potential hazards such as radioactive contamination, smoke, and temperature, etc. Pre-planned measures should be available to address these concerns for intentional and unintentional entry into the condition. The compensatory action may include placing fans in pull-to-lock as necessary to preclude there being a motive force to transport contaminated air to a clean environment in the event of an accident. These compensatory actions include administrative controls on opening plenums or other openings such that appropriate communication is established with the control room to assure timely closing of the system if necessary. Since the Fuel Handling Building boundary integrity also affects operability of the overall system, entry and exit is administratively controlled. Administrative control of entry and exit through doors is performed by the person(s) entering or exiting the area. Extended opening of the boundary is coordinated with the control room with appropriate plans for closure and communication.

3/4.7.9 (Not Used)

3/4.7.10 (Not Used)

3/4.7.11 (Not used)

3/4.7.12 (Not used)

3/4.7.13 (Not used)

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the Essential Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

When a risk-important system or component (for example Essential Chilled Water) is taken out of service, it is important to assure that the impact on plant risk of this and other equipment simultaneously taken out of service is assessed. The Configuration Risk Management Program evaluates the impact on plant risk of equipment out of service. A brief description of the Configuration Risk Management Program is in Section 6.8.3 (administration section) of the Technical Specifications.

The extended allowed outage time (EAOT) of 7 days for one inoperable Essential Chilled Water System loop is based on establishing compensatory measures that are consistent with the Configuration Risk Management Program and are controlled by plant procedures to offset the risk impacts of entering the EAOT. Refer to the Bases for 3.8.1.1. Action b for further details.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

- Maintenance in the switchyard that could directly cause a loss of offsite power is not allowed unless required to assure the continued reliability and availability of the offsite power
- Severe weather that could result in the extended loss of offsite power is not expected

Should one or more of these compensatory requirements not be met during the SDG out-of-service period, action will be taken in accordance with the CRMP to restore the function. If indicated by the risk assessment conducted in accordance with the program, other actions may be taken by station management to reduce risk by restoration of other components, rescheduling work that might increase the risk, or placing the unit in a more appropriate configuration.

If entry into the Action is unplanned (e.g., a failure of the SDG), station procedures require the implementation of the CRMP when the out-of-service time exceeds 72 hours. If one or more of the compensatory requirements is not functional at the end of the 72 hours, action will be taken in accordance with the CRMP to restore the function and to manage the risk.

TS 3.8.1.1 Action c.

To ensure a highly reliable power source remains with one offsite circuit and one diesel generator inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. However, if a second required circuit fails 4.8.1.1.1.a, the second offsite circuit is inoperable and LCO 3.0.3 should be entered. Action c provides an allowance to avoid unnecessary testing of OPERABLE diesel generators. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generators, and is an independently testable component or an inoperable support system, then surveillance requirement 4.8.1.1.2.a.2 does not have to be performed.

TS 3.8.1.1 Action d.

This action provides assurance that a loss of offsite power, during the period that a diesel generator is inoperable, does not result in a complete loss of safety function of critical systems. In this condition the remaining OPERABLE diesel generators and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may be lost; however, function has not been lost. Discovering one required diesel generator inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the operable diesel generator, results in starting the completion time for the required action. If the required number of channels or trains for a function or component is less than the total number of channels or trains and the TS allow unlimited operation with less than the total number of channels or trains (e.g. some Remote Shutdown System functions), then as long as there is emergency power for at least the required number of channels or trains, the requirements of TS 3.8.1.1.d are met. Similarly, if only one Reactor Containment Fan Cooler, out of six available, is inoperable, then there are no restrictions applied on the diesel generators and Action statement 3.8.1.1(d) (1) can be met.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

"...required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power" mean SSCs that are required by the Technical Specifications. TS 3.8.1.1.d. does not apply to non-TS SSCs that are governed by other documents (e.g. TRM).

The 24-hour completion time is based on the capability of the operable equipment to mitigate all but the most severe design basis accidents as described above and the extremely low probability of the occurrence of a design basis accident. The 24-hour completion time also allows a deliberate planned response that may allow the inoperable equipment to be restored.

TS 3.8.1.1 Action e.
Operation may continue for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources. With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient.

TS 3.8.1.1 Action f.

With two or three of the standby diesel generators inoperable, there is insufficient or no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. A single train onsite AC source can effectively mitigate all but the most severe events with operator action in some cases. The events that cannot be mitigated by a single train onsite AC source are highly unlikely. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Surveillance Requirements

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18. Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The Technical Specification Surveillance Requirements (SRs) for demonstrating the OPERABILITY of the standby diesel generators are in accordance with the recommendations of Regulatory Guide 1.108, Regulatory Guide 1.137, as addressed in the FSAR and NUREG-1431.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3744 is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1, allows for voltage drop to the terminals of 4000 V motors with minimum operating voltage specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of 4576 V corresponds to the 10% upper limit for the nominal 4160 volts on the safety bus.

The specified minimum and maximum frequencies of the standby diesel generators are 58.8 Hz and 61.2 Hz, respectively. These values are equal to plus or minus 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.108 and NUREG-1431.

SR 4.8.1.1.1.a

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution busses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 4.8.1.1.1.b

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that the components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 4.8.1.1.2.a.1

This SR provides verification that the level of fuel oil in the fuel tank is at or above the required level.

SR 4.8.1.1.2.a.2

This SR helps to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

## 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for $K_{eff}$ includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2800 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

### LCO 3.9.1.c

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in required boron concentration.

### BACKGROUND

During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

### APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves or mechanical joints during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves and mechanical joints are used to isolate unborated water sources. These devices have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.9.1 BORON CONCENTRATION (Continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

ACTIONS

The ACTIONS are modified to allow separate ACTION entry for each unborated water source isolation valve.

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve or mechanical joint required to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance the required action shall not preclude completion of movement of a component to a safe position.

ACTION b. includes a requirement that the verification that boron concentration is within limit be completed whenever ACTION b. is entered.

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation devices secured closed. Securing the valves or mechanical joints in the closed position ensures that the devices cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve or mechanical joint and secure the isolation device in the closed position immediately. Once actions are initiated, they must be continued until the devices are secured in the closed position.

Due to the potential of having diluted the boron concentration of the reactor coolant, verification of boron concentration per SR 4.9.1.2 must be performed whenever ACTION b. is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 4.9.1.3 These valves or mechanical joints are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under
SR 4.9.1.2. This Surveillance demonstrates that the devices are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.4.6
2. NUREG-0800, Section 15.4.6

# REPLACEMENT



## 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range and/or Extended Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

ACTION a. requires suspending the introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required refueling boron concentration limit. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

## 3/4.9.3 (Not Used)

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The containment personnel airlock and auxiliary airlock, which are part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation. The equipment hatch is required to be closed and sealed during MODES 1, 2, 3, and 4. During periods of shutdown, when containment closure is not required, the equipment hatch may be opened to allow passage of material needed to support activities in the containment building. The personnel and auxiliary airlock door interlock mechanisms may be disabled during shutdown, allowing both airlock doors to remain open for extended periods when frequent containment entry is necessary. Both containment personnel airlock doors may be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) there is 23 feet of water above the reactor vessel flange, (2) the reactor has been subcritical for ≥95 hours, (3) one airlock door is OPERABLE and, (4) an individual is available to close one personnel airlock door (if open) following a fuel handling accident inside containment.

The requirement to have 23 feet of water above the reactor vessel flange is consistent with the fuel handling accident analysis assumptions, Regulatory Guide 1.25, and Technical Specification 3.9.10, Water Level - Refueling Cavity.

Operability of a containment personnel airlock door requires that the door is capable of being closed, i.e., that the door is unblocked, no cables or hoses run through the personnel airlock, and at least one door seal is capable of being inflated. Containment personnel airlock door closure is required to take place within 30 minutes of initiation of a fuel handling accident inside containment if the reactor has been subcritical for less than 165 hours. Fuel movement is not permitted with personnel airlock doors open, if the reactor has not been subcritical for ≥95 hours. If the reactor has been subcritical for 165 hours or more, containment personnel airlock door closure is to occur as soon as practicable, but is assumed to occur within 2 hours to be consistent with the accident analysis.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (continued)

The equipment hatch may also be open during CORE ALTERATIONS when specific limitations are satisfied. The specification requires: (1) the reactor has been subcritical for ≥165 hours and, (2) the equipment hatch (if open) is capable of being closed following a fuel handling accident inside containment. The following administrative requirements will apply whenever the equipment hatch is open during core alterations or the movement of irradiated fuel in containment:

1. Appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS
2. Specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident
3. Obstructions (e.g., cables, hoses, and runway) that would prevent closure of the equipment hatch can be quickly removed.

The containment equipment hatch closure is required to take place upon the occurrence of a fuel handling accident inside containment if the hatch is open. Fuel movement is not permitted with equipment hatch open, if the reactor has not been subcritical for ≥165 hours. Equipment hatch closure should occur as soon as practicable, and is normally assumed to occur in 2 hours. Unlike the airlock, the equipment hatch may be blocked by an obstruction (e.g. the removable equipment hatch runway). Fuel movement is not allowed with the runway installed unless the capability to remove all obstructions and close the hatch within the required time is maintained.

A surveillance requirement verifies that the proper tools are staged at the equipment hatch location and qualified personnel assigned to close the equipment hatch on a seven-day frequency. These requirements assure that the associated doses are limited to within acceptable levels.

3/4.9.5 (Not Used)

3/4.9.6 (Not Used)

3/4.9.7 (Not Used)